Background

U.S.NRC
United States Nuclear Regulatory Commission
Protecting People and the Environment

References (1 of 3)

Web-based ADAMS Search: http://wba.nrc.gov:8080/ves/

Key Topic	Publication Type	Title	Reference
HF - Review	NUREG-1764, Rev 1	Guidance for the Review of Changes to Human Actions, Rev 1	ML072640413
HF - Review	NUREG-0711	Human Factors Engineering Program Review Model, Rev 2	ML040770540
HSI	NUREG-0700	Human-System Interface Design Review Guidleline: Review Methodology and Procedures. Vol. 1, Rev. 2	ML012120376
HSI	NUREG-0700	Human-System Interface Design Review Guidelines. Vol. 2, Rev 2, [1:3] Cover - Page 200,	ML021700337
HSI	NUREG-0700	Human-System Interface Design Review Guidelines. Vol. 2, Rev 2, [2:3] Page 201 - 400.	ML021700342
HSI	NUREG-0700	Human-System Interface Design Review Guidelines. Vol. 2, Rev 2, [3:3] Page 401 - End,	ML021700371
HSI	NUREG-0800, Appendix 18-A	Crediting Manual Operator Actions in Diversity and Defense-in-Depth (D3) Analyses, Rev 0, November 2009	ML092950353
HSI	NUREG-0800, Chapter 18.0	Standard Review Plan, Chapter 18.0 Human Factors Engineering, Rev 2, March 2007	ML070670253
HSI	NUREG-CR-5439	Human Factors Issues Associated with Advanced Instrumentation and Controls Technologies in Nuclear Plants.	ML071210310

References (2 of 3)

Key Topic	Publication Type	Title	Reference
HSI	NUREG-CR-5908	Advanced Human-System Interface Design Review Guideline: General Evaluation Model, Technical Development, And Guideline Description, Vol 1	ML071210321
HSI	NUREG-CR-5908	Advanced Human-System Inteface Design Review Guideline.. Evaluation Procedures And Guidelines For Human Factors Engineering Reviews, Vol 2	ML071220144
HSI - Adv Rx	NUREG-CR-6105	Human Factors Enginnering Guidance for the Review of Advanced Alarm Systems.	ML070460025
HSI - HCI	NUREG-CR-6633	Advanced Information Systems Design: Technical Basis and Human Factors Review Guidance	ML003704877
HSI - HCI	NUREG-CR-6947	Human Factors Considerations with Respect to Emerging Technology in Nuclear Power Plants.	ML083090338
Comp PROC	NUREG-CR-6634	Computer-Based Procedure Systems: Technical Basis and Human Factors Review Guidance	ML003704853
STAF & TRNG	NUREG-CR-6838	Technical Basis for Regulatory Guidance for Assessing Exemptions Requests from Nuclear Power Plant Licensed Operator Staffing Requirements Specified in 10CFR.54(m)	ML040580289

References (3 of 3)

Key Topic	Publication Type	Title	Reference
HRA	NUREG-1624	Technical Basis & Implementation Guidelines for a Technique for Human Event Analysis (ATHEANA), Rev 1, Pages 1-292	ML003719212
HRA	NUREG-1624	Technical Basis & Implementation Guidelines for a Technique for Human Event Analysis (ATHEANA), Rev 1, Appendices A - G	ML003719239
HRA	NUREG-1792	Good Practices for Implementing Human Reliability Analysis (HRA)	ML051160213
HRA	NUREG-1842	Evaluation of Human Reliability Analysis Methods Against Good Practices. Final Report	ML062630446
HRA	NUREG-1852	Demonstrating the Feasibilty and Reliability of Operator Manual Actions in Response to Fire, Final Report	ML073020676
HRA	NUREG-CR-6350	A Technique for Human Error Analysis (ATHEANA). Technical Basis And Methodology Description.	9607090121
HRA	NUREG-CR-6903, Vol. 1	Human Event Repository and Analysis (HERA) System: Overview	ML062700593
HRA	NUREG-CR-6903, Vol. 2	Human Event Repository and Analysis (HERA): The HERA Coding Manual and Quality Assurance	ML073130034

Human Reliability Analysis Model Differences

Background

The NRC's Office of Nuclear Regulatory Research (RES) is supporting the Advisory Committee on Reactor Safeguards (ACRS) to address the November 8, 2006, staff requirements memorandum (SRM-M061020) in which the Commission directed the ACRS to "work with the staff and other stakeholders to evaluate different human reliability models in an effort to propose a single model for the agency to use or guidance on which model(s) should be used in specific circumstances." RES is addressing this issue through collaborative work with EPRI (Electric Power Research Institute), initiated under the RES memorandum of understanding with EPRI on PRA.

Approach

To address the issue, the project is pursuing a formalization approach and a quantification tool capable of performing HRA in a consistent and efficient manner. The formalization approach aims to build a foundation for HRA that uses the current understanding of human performance and is consistent with the overall PRA framework from the perspective of both failure modeling and estimation of failure probabilities. This approach introduces the crew response tree (CRT) concept, which depicts human failure events in a manner parallel to the PRA event tree process. CRTs provide a structure for identifying the context associated with the human failure events under analysis and use a human information processing model as a platform to identify potential failures.

This approach incorporates behavioral science knowledge by providing the decompositions of human failures, failure mechanisms, and failure factors from both a top-down and bottom-up perspective. The bottom-up approach reflects findings from scientific papers documenting theories, models, and data of interest. The CRT structure and associated lower level models provide a roadmap for incorporating the phenomena with which crews would be dealing, the plant characteristics (e.g., design, indications, procedures, training), and the plant's human performance capabilities (understanding, decision, action). The work aims to create rules, and potentially template-based guidance, for a consistent, efficient, and effective analysis.

For quantification, the formalization approach uses the typical PRA conditional probability expression, delineated to a level adequate for associating the probability of a human failure event with conditional probabilities of the associated contexts, failure mechanisms, and underlying factors (e.g., performance-shaping factors). This mathematical formulation can be used to directly estimate HEPs using various data sources (e.g., expert estimations, anchor values, simulator data, historical data) or can be modified to interface with existing quantification approaches. However, the quantification approach is still under exploration.

The staff anticipates that the methodology will be developed and available for public review and comment by November 2011.

The NRC's costs represent only a fraction of the actual costs for both the international empirical study and the collaborative work with EPRI for addressing SRM-M061020 on HRA model differences. Through these collaborative efforts, the NRC is also able to take advantage of extensive domestic and international PRA and HRA expertise from recognized academics and practitioners.

For More Information

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Improving Human Reliability Analysis Methods by Using Simulator Runs

Background

As part of its efforts to improve human reliability analysis (HRA), the Office of Nuclear Regulatory Research (RES) participates in and supports the International HRA Empirical Study to benchmark HRA models by comparing HRA results to empirical data generated through crew simulator runs. Although the documentation of this study is not yet complete, its findings indicate areas for improvement in HRA methods and practices. But because the study is based on the results of simulator runs using European crews at the Halden Reactor Project (HRP) simulator, the issue of the applicability of the study results to U.S. nuclear power plant crews has been raised.

In its February 2009 staff requirements memorandum (SRM-M090204B), the Commission directed the staff to work with industry and international partners to test the performance of U.S. nuclear power plant operating crews and to keep the Commission informed of the status of its HRA data and benchmarking projects. RES's benchmarking work is responsive to SRM-M090204B.

The NRC established a memorandum of understanding with a U.S. utility that volunteered to participate in this study and offered simulator facilities, crews, and expertise to support the design and execution of the experimental runs. As a result, a new study was initiated that the HRP staff supports with expertise in the design and execution of simulator runs, as well as the collection and interpretation of crew performance data.

The objective of this new study is to evaluate a specific set of HRA methods used in regulatory applications by comparing HRA predictions to crew performance in simulator experiments performed at a U.S. nuclear power plant. The results will be used to accomplish the following:

- Determine the potential limitations of data collected in non-U.S. simulators when used to evaluate U.S. applications.
- Improve the insights developed from the International HRA Empirical Study.

Approach

The study approach consists of the following four steps:

1. Experimental Design and Performance of Simulated Scenarios

The experimental design is focused on collecting information on the predictive power and consistency of HRA methods—A Technique for Human Even Analysis (ATHEANA), Standardized Plant Analysis Risk—Human Reliability Analysis Method (SPAR-H), Technique for Human Error Rate Prediction/ Accident Sequence Evaluation Program (THERP/ASEP), and Cause-Based Decision Tree (CBDT) in particular—through analysis of crew performance in simulated nuclear power plant accident scenarios. It stipulates the collection of information to be used by HRA analysts to evaluate the human failure events (HFEs) involved in the scenarios and to estimate the human error probabilities (HEPs).

The design includes three accident scenarios. The design addresses the plant status before the initiating event, the initiating event, and the associated plant design capabilities and operational characteristics to deal with the event, including procedural guidance; the predetermination and definition of the HFEs to be analyzed for each scenario and associated success criteria; the identification of human performance metrics; the development of crew performance collection protocols and questionnaires to support documentation of observed crew performance; and the development of an information package containing basic probabilistic risk assessment (PRA) and HRA information to be provided to the HRA teams.

The actual experiment consists of the running of the scenarios and the collection and documentation of observations about plant behavior and crew performance by experts (typically plant trainers and PRA/HRA experts). In addition to live observations, crew performance observations are collected through videotapes and debriefings of both the crews and the plant experts who observed the runs.

The experimenters evaluate crew performance by analyzing the information collected during the experiment according to predefined protocols and performance metrics. This part of the study is supported by the staff of the HRP.

2. Information Collection and Evaluation of HEPs by HRA teams

Each HRA method is applied by two or three HRA teams composed of NRC and contractor staff. The HRA teams visit the plant to interview plant personnel, view simulator runs (other than the study simulations), and collect relevant plant information. On the basis of the information collected, the teams use their selected HRA methods to perform predictive analysis and to estimate HEPs for the HFEs involved in the simulated scenarios, document the results, and submit them for review and evaluation.

One goal of the study is to understand the types of information considered by HRA teams in performing HRA analysis using a given method. Documenting this information provides insights about differences and commonalities among HRA methods; in particular, it helps staff to develop an understanding of how methods (or analysts) are using the collected information and of how the different ways of using information affect consistency among methods or analysts. Documenting information use also allows comparisons with crew simulator performance to examine if the appropriate factors are being considered by the teams using the different HRA methods.

3. Evaluation of the HRA submittals

An independent group of experts reviews the submitted analyses and compares them to the observed simulator data. These experts perform method-to-method and HRA team-to-team comparisons to determine if and how method differences and analyst differences influence the HRA results. Their analysis includes both qualitative and quantitative comparisons.

Qualitative comparisons examine the extent to which HRA analysts, using their methods, were able to identify key drivers (such as misdiagnosis of equipment failures or lack of adequate procedural guidance for performing the required actions) that could influence the crew's capability to accomplish the required actions. Through such comparisons, the experts identify (1) method limitations with regard to guiding analysts to identify important drivers of human performance, and (2) method limitations with regard to ensuring a consistent use of the method by different analysts (intra-analyst consistency).

Quantitative comparisons involve (1) the ranking of the estimated HEPs, (2) the ranking of the human actions in terms of the level of difficulty that crews appear to have experienced during the simulation, and (3) comparison of the resulting ranking in (1) and (2). These comparisons allow the experts to examine whether or not inconsistencies in ranking stem from the following causes:

- the extent to which the quantification tool can incorporate the important drivers of human performance identified through the qualitative analysis (e.g., the tool allows the use of only a few performance shaping factors in the estimation of HEPs)
- the extent to which the quantification tool can provide a consistent and traceable process to estimate HEPs
- the analysts' capability to correctly apply the tool.

4. Documentation of the Results

A NUREG report will (1) document the results for each method tested, including the performance characteristics of each method and potential implications for regulatory applications, and (2) assess the consistency of the methods and identify how practitioners can achieve better consistency in HRA.

RES expects this study to be completed by September 2011.

For More Information

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Pilot Testing of Human Reliability Analysis-Informed Training and Job Aid for NRC Staff Involved with Medical Applications of Byproduct Materials

Background

In 2003, the Office of Nuclear Material Safety and Safeguards (NMSS) provided the Office of Nuclear Regulatory Research (RES) with a user need for developing human reliability analysis (HRA) capability specific to materials and waste applications (NMSS-2003-003). In this memorandum, NMSS requested two phases of work. Both phases were completed in December 2008.

The Phase 1 work consisted of feasibility studies for developing NMSS capability in HRA. The feasibility study for materials applications addressed both medical and industrial applications.

The Phase 2 work focused on the recommendations from the feasibility study, namely, the development of job aids (e.g., HRA-informed decisionmaking aids) and associated training for NRC staff on HRA-informed issues in human performance in medical applications.

The final products of the Phase 2 work, a prototype HRA-informed job aid (i.e., a database of risk-relevant human performance issues and historical errors, related to treatment steps) and associated training materials for medical applications (gamma-knife based), were presented to staff in the Office of Federal and State Materials and Environmental Management Programs (FSME) and delivered to the NRC in December 2008.

Follow-up work to pilot the HRA-informed job aid and training materials began in spring 2010.

Approach

The overall objective is to develop HRA-informed job aids and associated training for NRC staff involved with medical applications of byproduct materials. Although prototypes of the HRA-informed job aid and training materials have been developed, instructions on how to use these tools for specific NRC tasks (e.g., inspections, license reviews) were not developed. Consequently, interaction with NRC staff from the regions, as well as the continued involvement of staff at NRC Headquarters is required in this pilot testing phase of development.

RES is currently making plans for pilot testing of both products at NRC Region I.

PILOT TESTING TASKS

The following are the expected tasks for the pilot testing of the HRA-informed job aid and associated training:

- initial updates to HRA-informed training and job aid (with respect to recent events and new gamma knife technology)
- initial interactions with NRC Region I staff
- onsite HRA-informed training
- onsite demonstration of HRA-informed job aid
- selection of candidates for trial use of HRA-informed job aid
- trial use of HRA-informed job aid
- feedback on trial use
- updates to HRA-informed job aid and associated training (based on feedback)

By the end of calendar year 2010, the first two tasks should be complete and preparations started for Task 3.

For More Information

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Qualitative Human Reliability Analysis for Spent Fuel Handling

Background

In 2003, the Office of Nuclear Material Safety and Safeguards (NMSS) provided the Office of Nuclear Regulatory Research (RES) with a user need for developing human reliability analysis (HRA) capability specific to materials and waste applications (NMSS-2003-003). In this memorandum, NMSS requested two phases of work, the first of which is completed.

Phase 1 work consisted of feasibility studies for developing NMSS capability in HRA. The feasibility study for waste applications (performed by NRC staff) addressed high-level waste, spent fuel storage, fuel cycle, and decommissioning applications. This study identified the following needs for potential NMSS-specific HRA development that were common to more than one waste application:

- development of HRA methods specific to NMSS needs
- guidance for evaluating the effectiveness of administrative controls
- guidance on good practices for implementing HRA
- guidance for reviewing HRAs
- assistance in incident significance assessments

Initial Phase 2 work on this project began investigating development of HRA methods specific to NMSS needs and guidance on good practices for implementing HRA.

Additionally, NMSS and RES identified new priorities, resulting in project efforts focused on the development of HRA insights for spent fuel handling. Such activities include investigation of both spent fuel misloads and cask drops.

Approach

The first step in developing HRA capability for NMSS was to develop a qualitative understanding of the important human performance issues for spent fuel handling that need to be addressed by HRA.

To this end, this project has completed the following work:

- identification and review of literature relevant to understanding human performance in spent fuel handling
- interviews of subject-matter experts in spent fuel handling
- evaluation and use of relevant literature and interviews of experts to perform qualitative HRA tasks for spent fuel handling

The result of this work was a July 2006 Sandia National Laboratories (SNL) letter report describing potential vulnerabilities and possible scenarios that could lead to misloads and cask drops.

Currently, the project is developing further HRA-informed insights on cask drops. It is expected that this work will provide useful input to future NRC inspections and reviews.

The current schedule for deliverables for both efforts is the following:

- draft NUREG/CR on initial efforts for misloads and cask drops—September 2010
- draft NUREG/CR on recent expanded efforts on cask drops—September 2010
- preparation of both final NUREG/CRs—December 2010

Continued interactions between NMSS and RES staff are planned as these deliverables are completed.

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Human Performance for Advanced Control Room Design

Background

With the renewed interest in nuclear energy, there are plans to begin constructing new plants within the next several years. The new generation of plants will differ from the existing fleet in several important ways, including the design of the reactors, the instrumentation and control (I&C), and the human-system interface (HSI). Figure 6.2 illustrates one conceptualization of an advanced control room (CR) design. Taken together, these technological advances will lead to concepts of operation that are different from those found in currently operating nuclear power plants. The potential benefits of the new technologies should result in more efficient operations and maintenance. However, if the technologies are poorly designed and implemented, there is the potential they will reduce human reliability, increase errors, and negatively impact human performance—resulting in a detrimental effect on safety. To address these concerns, the NRC sponsored a study to identify human performance research that may be needed to support the review of licensee's implementation of new technology in new and advanced nuclear power plants.

Approach

To identify the research issues, current industry trends and developments were evaluated in the areas of reactor technology, I&C technology, HSI integration technology, and human factors engineering (HFE) methods and tools. These four research issues were then organized into seven HFE topic areas: (1) role of personnel and automation, (2) staffing and training, (3) normal operations management, (4) disturbance and emergency management, (5) maintenance and change management, (6) plant design and construction, and (7) HFE methods and tools. Next, a panel of independent subject-matter experts representing various disciplines (e.g., HFE, I&C) and backgrounds (e.g., vendors, utilities, research organizations) prioritized the issues. Sixty-four issues were distributed among four categories, with 20 research issues placed into the top priority category.

NUREG/CR-6947, "Human Factors Considerations with Respect to Emerging Technology in Nuclear Power Plants," issued October 2008, documents the results of the study. The report contains a summary of the high-level topic areas, the research issues in each topic area, the priorities for each issue, and a human performance rationale that describes the reason why each research issue is relevant. The findings from this study are being used to develop a long-term research plan addressing human performance within these technology areas for the purpose of establishing a technical basis from which regulatory review guidance can be generated.

Of the 20 research projects identified as having a priority 1 research need, four have been completed, five are currently underway, and an additional three projects are scheduled to begin this year. Descriptions of the five projects that are underway are provided below.

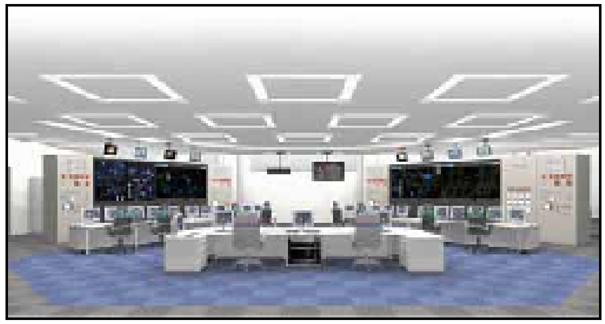

Figure 6.2 One conceptualization of an advanced control room design

Advances in Human Factors Engineering Methods and Tools

The methods and tools used to design, analyze, and evaluate the HFE aspects of nuclear power plants are rapidly changing. A previous study identified the current trends in the use of HFE methodologies and tools, identified their applicability to nuclear power plant design and evaluation, and determined their role in safety reviews conducted by the NRC. The study identified seven categories of methods and tools for which additional review guidance may be needed, including (1) application of human performance models, (2) use of virtual environments and visualizations, (3) analysis of cognitive tasks, (4) rapid development engineering, (5) integration of HFE methods and tools, (6) computer-aided design, and (7) computer applications for performing traditional analyses. One outcome of this project to date has been the development of detailed guidance for applying human performance models to the evaluation of nuclear power plant designs. The next phase of the study will provide human factors (HF) guidance for an additional two methods and tool categories.

Roles of Automation and Complexity in Control Rooms

The overall level of automation in advanced nuclear power plants is expected to be much higher than in plants currently operating in the United States. It is important that the staff be cognizant of current practices and trends in the use of automation in nuclear power plant CRs and understand the influences of automation on CR design, human performance, and conduct of operations. A previous study, "Human-System Interfaces (HSIs) to Automatic Systems," developed a general framework for characterizing automation systems and developed HFE criteria for evaluating automation designs. The present study will further the state of the art by examining the impact of automation on CR design, specifically the impact of automation on (1) operator performance during normal, abnormal, and emergency operations; (2) the reliability of operator's use of automation systems, including existing methods for assessing impacts; and (3) operator performance when the automation fails or is in a degraded state.

Human Factors Guidance for the Assessment of Computerized Procedures

Applicants for new and advanced reactor design certifications are proposing to incorporate computer-based procedure capabilities as part of their main CR designs. The potential forms of implementation can range from basic applications that are limited to displaying static representations of procedures to those that provide dynamic displays of procedures in conjunction with relevant plant status and process data, context-dependent decision aids, soft controls, and the capability to implement automated sequences of procedure steps. Although the challenges and human factors considerations increase with the level of functionality of these applications, even the most basic application requires consideration of how it will be integrated with other elements of the CR design, how the implementation might affect the roles and responsibilities of the operating crew and standards for conduct of operations, how the operators will transition to backup procedures upon loss of a computer-based procedure system, what the potential failure modes of the application will be, and how those failure modes will be addressed to ensure that acceptable levels of human performance will be maintained. This project will review applicable research literature and operating experience and develop a technical basis document for the development of review guidance that addresses the key issues associated with the use of CR computerized procedures.

Human Factors Aspects in Concepts of Operations for Modular Designs

Advances in nuclear power plant technology have set the stage for changes to traditional concepts of operations (CONOPS). The CONOPS of new reactor designs introduce such safety-critical performance considerations as the operation of multiple reactors by a reduced crew. The objective of this project is to examine the human factors aspects associated with the monitoring and control of multimodular plants and to provide a technical basis for evaluating the impacts of evolving CONOPS on human performance. The regulatory documents for reviewing modular designs will also be assessed to identify areas that need additional technical basis or guidance to facilitate the staff review of CONOPS for modular reactor designs.

Update Existing Human Factors Engineering Regulatory Guidance

The NRC staff reviews the HFE aspects of nuclear power plants in accordance with the guidance presented in NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants." Detailed design review procedures for the HFE programs of applicants for construction permits, operating licenses, standard design certifications, combined operating licenses, and license amendments are provided in NUREG-0711, Revision 2, "Human Factors Engineering Program Review Mode." As part of the review process, the interfaces between plant personnel and plant systems and components are evaluated for conformance with the guidance contained in NUREG-0700, Revision 2, "Human-System Interface Design Review Guidelines." NUREG-0711 and NUREG-0700 were last updated in 2004 and 2002, respectively. This study will update NUREG-0711 and NUREG-0700 with HFE criteria developed from the most recent and best available technical bases. The availability of up-to-date HFE review guidance will help to ensure that the NRC staff has the latest knowledge, information, and tools to safely and efficiently perform its regulatory tasks.

For More Information

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Support For Implementation of 10 CFR Part 26 Fitness-for-Duty Programs

Background

To ensure the safety and security of nuclear facilities, the NRC has developed regulations to standardize and ensure effective implementation of fitness-for-duty (FFD) programs that apply to personnel who engage in certain safety- and security-related activities. For example, certain personnel at commercial nuclear power plants who have unescorted access to the plant's protected areas and those who transport strategic special nuclear materials must be subject to an FFD program. The NRC requires FFD programs to provide reasonable assurance that nuclear facility personnel are trustworthy and will perform their tasks in a reliable manner.

In Title 10 of the *Code of Federal Regulations* (10 CFR) Part 26, "Fitness for Duty Programs," the NRC describes the scientific and technical requirements for FFD programs that address illegal drug use, alcohol abuse, misuse of legal drugs, impairment from fatigue, and any other mental or physical conditions that could impair job performance. At the time that 10 CFR Part 26 was first published in the *Federal Register* (54 FR 24468; June 7, 1989) and subsequently, the Commission directed the NRC staff to continue to analyze FFD programs, assess the effectiveness and efficiency of the rule, and recommend appropriate improvements or changes.

Most recently, the NRC, with extensive stakeholder input, published an amended, reorganized, and updated rule. The amended 10 CFR Part 26 was published in the *Federal Register* on March 31, 2008. It is organized into 12 subparts that group together related requirements. The NRC permitted licensees and other entities to defer implementation of the majority of the rule's requirements until March 31, 2009, and granted an additional 6 months to implement the rule's new fatigue management requirements.

Approach

The Office of Nuclear Regulatory Research (RES) participates in a multidisciplinary team of NRC staff that is supporting a myriad of agency initiatives and efforts to facilitate education about the rule and its implementation.

Fatigue Regulatory Guide

RES worked closely with other NRC staff and stakeholders to publish guidance for implementing the fatigue management requirements of 10 CFR Part 26. Specific requirements for nuclear power plant licensees to manage worker fatigue are a new addition to 10 CFR Part 26. As guidance on the new rules, the NRC published Regulatory Guide (RG) 5.73, "Fatigue Management for Nuclear Power Plant Personnel," in March 2009.

Training Development

To ensure that implementation efforts among the regions and various offices are coordinated and consistent, RES staff and its contractors have developed training materials for inspectors and other NRC staff involved in implementing 10 CFR Part 26. To date, the training has been developed, pilot-tested, and supplemented with computer-based training specifically focused on the fatigue management requirements.

FFD Web Site Update

Transparency is an important NRC goal. Toward that end, the NRC staff maintains a public Web site to provide one location for stakeholders to access information and submit questions about the rule and any implementation concerns. The Web site includes the history of the 10 CFR Part 26 rulemaking, frequently asked questions about 10 CFR Part 26 and its implementation, FFD program reports from licensees, and related documents and resources.

Inspection Procedures

RES supports other NRC offices in developing inspection procedures that are used to evaluate the effectiveness of FFD programs and to verify licensee compliance with the rule's requirements.

Technical Bases for Alternate Specimens and Fatigue Technologies

The science and technologies for assuring personnel fitness for duty continue to advance. Consistent with the Commission's direction to continue assessing the effectiveness and efficiency of FFD programs, RES is identifying scientific and technological advances that may enhance FFD programs. For example, 10 CFR Part 26 currently requires the use of urine, breath, and saliva testing for drugs and alcohol. However, new drug testing technologies are being developed that rely on alternate specimens, including hair and sweat. New methods to manage fatigue in the workplace and technologies for assessing fatigue and other possible types of impairment are also of interest. Finally, RES is evaluating other readiness-to-perform technologies, as these tests have implications for effective job and task performance.

Future Updates to 10 CFR Part 26

The Commission directed the NRC staff to initiate a new 10 CFR Part 26 rulemaking after publication of the March 31, 2008, amended and revised rule. The Commission asked the NRC staff to review specific elements of the rule related to the technical basis and to evaluate including licensee quality control, quality verification, and quality assurance personnel in the fatigue provisions of 10 CFR Part 26. The RES staff is continuing to provide its technical expertise to staff engaged in the new rulemaking.

For More Information

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Safety Culture

Background

The culture of an organization affects the performance of the people in it. Weaknesses in an organization's safety culture may set the stage for equipment failures and human errors that can have an adverse impact on safe performance.

Goal of Safety Culture Activities

The initial goal of the NRC's 2006 safety culture initiative was to enhance the Reactor Oversight Process (ROP) to more fully consider safety culture in the NRC's assessments of inspection findings and overall nuclear power plant performance. More recently, the Commission directed the NRC staff to (1) consider the need for an agencywide safety culture policy statement that would apply to all entities regulated by the NRC and (2) recommend whether and how to better integrate security culture considerations into the NRC's safety and security oversight activities.

The Office of Nuclear Regulatory Research (RES) is providing technical expertise related to human and organizational performance to support the agency's safety culture activities. The RES staff participates in the Safety Culture Working Group, the Safety Culture Policy Statement Task Force, and the Safety Culture Policy Statement Steering Committee.

Industry Safety Culture Assessment Initiative

Concurrent with the NRC staff's activities, the nuclear power industry, led by the Nuclear Energy Institute (NEI), is developing a standardized safety culture assessment methodology and performance indicators. NEI has indicated that the assessment methodology will be used by nuclear power plant licensees for biennial self-assessments and, with modifications, to reply to NRC requests for independent or third-party safety culture assessments under the ROP. The performance indicators will be used to provide ongoing monitoring of safety culture trends. RES staff will assist the Office of Nuclear Reactor Regulation (NRR) in evaluating the industry's new approach.

For More Information

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Fire Human Reliability Analysis Methods Development

Background

The Individual Plant Examination of External Events (IPEEE) program and the experience from actual fire events found that, depending on design and operational conditions, fire can be a significant or dominant contributor to nuclear power plant (NPP) risk. Human errors have been shown to be a significant contributor to overall plant risk (including the risk from fires) because of the significant role that operators play in the fire protection strategy on reactor safety. Figure 7.2 illustrates operators in an NPP control room. Human reliability analysis (HRA) is the tool used to assess the implications of various aspects of human performance on risk. Currently, the NRC is expanding existing HRA methods to evaluate the impact of human failures in the fire protection defense-in-depth safety strategy.

In 2004, the NRC amended its fire protection requirements to allow existing reactor licensees to voluntarily adopt the risk-informed, performance-based rule, 10 CFR 50.48(c). This rule endorses National Fire Protection Association (NFPA) Standard 805, "Performance Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants," as an alternative to the existing prescriptive fire protection requirements. To realize the full benefits of making the transition to the risk-informed, performance-based standard, plants will need to have a fire probabilistic risk assessment (PRA) that includes quantitative HRA for post-fire mitigative human actions modeled in a fire PRA.

The Electric Power Research Institute (EPRI) and NRC's Office of Nuclear Regulatory Research (RES) embarked on a cooperative project to improve the state of the art in fire risk assessment to support this new risk-informed environment in fire protection. This project produced a consensus document, NUREG/CR-6850 (EPRI 1011989), "Fire PRA Methodology for Nuclear Power Facilities," that addresses fire risk for at-power operations. This report provides high-level qualitative guidance and quantitative screening guidance for conducting a fire HRA. However, this document does not provide a detailed quantitative methodology to develop best-estimate human error probabilities (HEPs) for human failure events under fire-generated conditions.

Objective

The overall objective of the effort is to develop fire HRA methods beyond what is currently in NUREG/CR-6850 (EPRI 1011989) and develop an HRA methodology and approach suitable for use in a fire PRA.

The fire HRA guidance developed through this effort is intended to support plants making the transition to 10 CFR 50.48(c), as well as NRC reviewers evaluating the adequacy of submittals from licensees making that transition. It may also be employed as a general fire PRA tool for HRA.

Figure 7.2 Operators in a NPP control room

Approach

RES has worked collaboratively with EPRI to develop a methodology and associated guidance for performing quantitative HRAs for post-fire mitigative human actions modeled in a fire PRA. The NRC issued NUREG-1921 (EPRI 1019196), "EPRI/NRC-RES Fire Human Reliability Analysis Guidelines" (see Figure 7.3), as a draft for public comment in December 2009. It provides three approaches to quantification: screening, scoping, and detailed HRA. Screening is based on the guidance in NUREG/CR-6850 (EPRI 1011989), with some additional guidance for scenarios with long time windows. Scoping is a new approach to quantification developed specifically to support the iterative nature of fire PRA quantification. Scoping is intended to provide less conservative HEPs than screening but requires fewer resources than a detailed HRA. For detailed HRA quantification, the NRC has developed guidance on how to apply existing methods to assess post-fire HEPs.

The NRC plans to release NUREG-1921 (EPRI 1019196) as a final report in spring 2011.

Future Work

The NRC has added a new HRA module to the NRC-RES/EPRI Fire PRA Workshop to provide training on the use of this methodology. The joint fire HRA methodology development team is scheduled to deliver the fire HRA training at the 2010 workshops, as well as at future fire PRA workshops.

Figure 7.3 NUREG-1921 cover page

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The Organization for Economic Cooperation and Development Halden Reactor Project

Background

The NRC and its predecessor, the U.S. Atomic Energy Commission (AEC), have been participating in the Organization for Economic Cooperation and Development (OECD) Nuclear Energy Agency (NEA) Halden Reactor Project (HRP) since its inception in 1958. During this period, the NRC has used numerous research products from this internationally funded cooperative effort, which is located in Halden, Norway, and managed by the Norwegian Institute for Energy Technology (Institutt for Energiteknikk (IFE). For example, Halden tests on high-burnup fuel under loss-of-coolant accident (LOCA) conditions supported an NRC research information letter on cladding embrittlement. As another example, Halden's human factors research has supported regulatory guidance in areas such as alarm systems, hybrid control rooms, display navigation, and guidance for the review of proposed staffing configurations in computer-based control rooms.

Facilities and Activities

Fuels and Materials Research

The Halden Boiling-Water Reactor (HBWR) (see Figure 12.1), which currently operates at 18 to 20 megawatts, is fully dedicated to instrumented in-reactor testing of fuel and reactor materials. Since its initial startup, the reactor facility has been progressively updated and is now one of the most versatile test reactors in the world. The HRP fuels and materials program focuses on the performance of fuel and structural materials under normal or accident conditions using the numerous experimental channels in the core that are capable of handling many test rigs simultaneously.

Recent NRC reviews of industry fuel behavior codes have directly employed data from the HRP fuels program. These data are also essential for updating the NRC's fuel codes and materials properties library, which are used to audit industry analyses. Currently, the NRC is particularly interested in the previously mentioned LOCA tests, which are investigating such phemonena as axial gas flow, maintaining or breaking fuel-to-cladding bonding, fuel axial relocation, and fuel fragment spillage through cladding burst opening.

Regarding the HRP's nuclear reactor materials testing program, the HRP has, over the years, provided fundamental technical information to support the understanding of the performance of irradiated reactor pressure vessel (RPV) materials and supplemented results generated under NRC research programs. Recently, the HRP has been an essential partner in evaluating the irradiation-assisted stress-corrosion cracking (IASCC) of light-water reactor (LWR) materials. The HRP has irradiated materials that were later tested under the NRC's research program at Argonne National Laboratory to measure crack initiation, fracture toughness, and crack growth rate under representative LWR conditions. The HRP's ongoing work on IASCC and other areas (e.g., irradiation-induced stress relaxation) supplements NRC-sponsored research and addresses existing knowledge gaps. The NRC staff is using this information to inform reviews of licensee aging management programs.

Man-Technology-Organization Laboratory

IFE's Halden facility also includes the IFE Man-Technology-Organization (MTO) Laboratory. The Halden Man-Machine Laboratory (HAMMLAB) (see Figure 12.2) is one of the principal experimental facilities in this laboratory. HAMMLAB uses a reconfigurable simulator control room that facilitates research into instrumentation and control (I&C), human factors, and human reliability analysis (HRA). Currently, HAMMLAB has hardware and software enabling it to simulate the Fessenheim pressurized-water reactor (PWR) plant in France, the Forsmark-3 boiling-water reactor (BWR) plant in Sweden, and the Ringhals-3 PWR plant in Sweden.

Many of the HAMMLAB experiments are performed with the control room configured as a prototype advanced control room with an integrated surveillance and control system. This setup is used to explore the impacts of automation and advanced human-system interfaces on operator performance. HAMMLAB has extensive data collection capabilities and typically uses qualified nuclear power plant operators (who are familiar with the plants being simulated) as test subjects.

Recently, HRP-designed and executed HAMMLAB experiments provided the foundation for the International Empirical HRA Study, a multinational study aimed at developing an empirically based understanding of the performance, strengths, and weaknesses of HRA methods used in risk-informed regulatory applications. The NRC will be using the study's results to address outstanding HRA technical issues, including those related to HRA model differences identified in a November 8, 2006, staff requirements memorandum (SRM). Currently, ongoing HRP experiments are addressing a number of topics of interest to the NRC, including control room staffing strategies, the role and effects of automation in advanced control room designs, and aids to improve control room teamwork. The NRC expects that this research will contribute to the technical basis for human factors guidance, especially for new reactor designs.

The IFE MTO Laboratory also includes a virtual environment center and an integrated operations laboratory. The former is used to perform research involving mixed reality applications (e.g., training), and the latter is used to address issues associated with remote operations.

Finally, the MTO Laboratory also conducts research on I&C systems. Past efforts include work in the area of instrumentation surveillance and monitoring techniques based on advanced decision algorithms. A number of HRP-developed systems have been evaluated for use by U.S. plants.

The current HRP digital systems research activities contribute to three phases of a system lifecycle:

- Development, assurance and deployment of high integrity software important to nuclear power plant safety,
- Condition monitoring and maintenance support, where engineering and technical support teams are the intended beneficiaries of the research results. This research will improve accuracy and usability of current methods and develop novel techniques to improve diagnostics and condition-based maintenance.
- Development and application of software systems for operational support, where plant operators are the intended beneficiaries of the research results. The research program includes interaction of advanced control systems with human operators and issues related to the implementation and use of operational procedures.

Summary

The HRP has provided and continues to provide valuable information to the NRC. Much of this information addresses gaps that are otherwise not being addressed by current NRC research activities, and some of this information is foundational to NRC's efforts to improve the technical basis of key models, methods, and tools. Furthermore, because the NRC is one of several contributors to the HRP budget, the HRP enables the NRC staff to significantly leverage its resources.

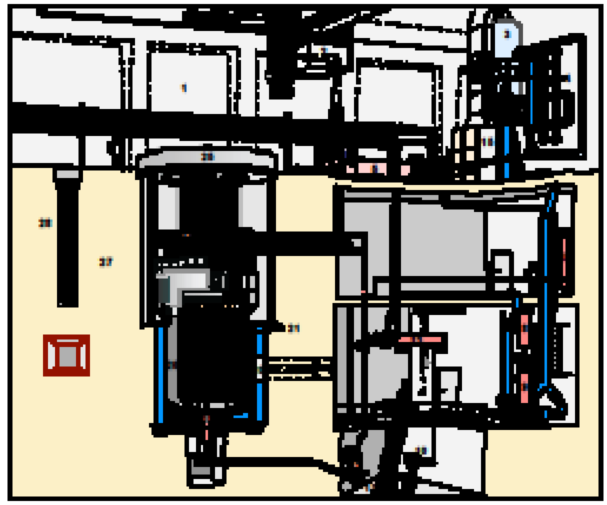

Figure 12.1 HBWR test reactor

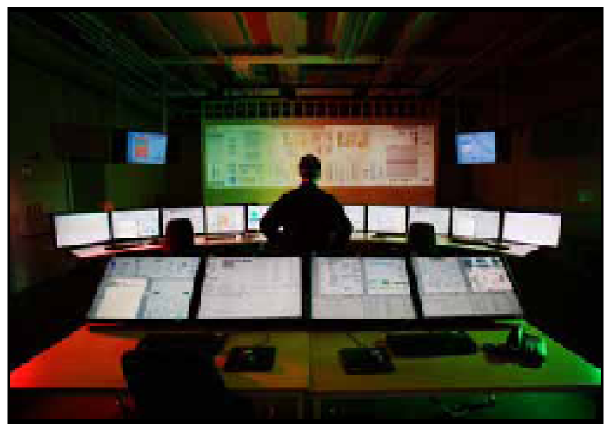

Figure 12.2 HAMMLAB control room simulator

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